November 9, 2004

Mr. William Pearce
Vice President
FirstEnergy Nuclear Operating Company
Post Office Box 4
Shippingport, Pennsylvania 15077

SUBJECT: BEAVER VALLEY POWER STATION - NRC INTEGRATED INSPECTION REPORT 05000334/2004005 AND 05000412/2004005

Dear Mr. Pearce:

On September 30, 2004, the United States Nuclear Regulatory Commission (NRC) completed an inspection at your Beaver Valley Power Station Units 1 and 2. The enclosed integrated inspection report documents the inspection findings, which were discussed on October 25, 2004, with you and other members of your staff.

The inspections examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, this report documents three NRC-identified findings of very low significance (Green). Two of these findings were determined to be violations of regulatory requirements. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these two findings as non-cited violations consistent with Section VI.A of the NRC Enforcement Policy. If you contest any aspect of this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Beaver Valley Power Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, and its enclosures, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

We appreciate your cooperation. Please contact me at 610-337-5234 if you have any questions regarding this letter.

Sincerely,

/RA/

Peter W. Eselgroth, Chief
Reactor Projects Branch 7
Division of Reactor Projects

Docket Nos.: 50-334, 50-412
License Nos: DPR-66, NPF-73

Enclosures: Inspection Report 05000334/2004005; 05000412/2004005
w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos.	50-334, 50-412
License Nos.	DPR-66, NPF-73
Report Nos.	05000334/2004005 and 05000412/2004005
Licensee:	First Energy Nuclear Operating Company (FENOC)
Facility:	Beaver Valley Power Station, Units 1 and 2
Location:	Post Office Box 4 Shippingport, PA 15077
Dates:	July 01, 2004 - September 30, 2004
Inspectors:	P. Cataldo, Senior Resident Inspector G. Smith, Resident Inspector D. Merzke, Resident Inspector T. Fish, Sr. Operations Engineer L. Scholl, Senior Reactor Inspector
Approved by:	P. Eselgroth, Chief Reactor Projects Branch 7 Division of Reactor Projects

CONTENTS

SUMMARY OF FINDINGS

IR 05000334/2004005, IR 05000412/2004005; 07/01/2004 - 09/30/2004; Beaver Valley Power Station, Units 1 & 2; Flood Protection Measures; Licensed Operator Requalification Program; Operability Evaluations.

The report covered a 3-month period of inspection by resident inspectors and announced inspections by two regional inspectors. Two Green non-cited violations (NCVs), and one Green Finding were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609 "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be "Green" or be assigned a severity level after Nuclear Regulatory Commission (NRC) management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Rev. 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a non-cited violation of 10 CFR 50 Appendix B, Criterion XI, "Test Control," because four Unit 1 flood control level switches were found inoperable. Specifically, the NRC identified that these flood switches lacked surveillance or functional testing requirements that would have identified the inoperable and poorly designed flood switches.

  The finding is more than minor since it affects the reliability of various mitigating systems during a flooding scenario. If an internal or external flood had occurred, no alarm would be received in the control room. The finding is of very low safety significance because operator rounds each shift would promptly alert the control room personnel of flooding conditions which could affect mitigating systems (Section 1R06).

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of Technical Specification 6.8.1, because an improperly built scaffold adversely impacted the Unit 2 'C' service water pump. Specifically, a scaffold bar was attached to the motor lifting lug of the service water pump, contrary to the scaffold erection procedure.

  The finding is more than minor because it adversely affected the reliability of a safety-related service water pump as well as the mitigating systems cornerstone objective. The finding is of very low safety significance since further engineering analysis determined that the pump would have remained operable, and therefore capable of performing its design basis function (Section 1R15).

- Green. The inspectors identified a finding in that the licensee's methodology for simulator testing deviated from the accepted guidance; the potential existed for deviations to be introduced between the plant control room and the plant reference simulators and deviations were consequently identified that could cause negative training, which in turn could have an adverse effect on operator actions during plant operations.

The finding is more than minor because it affects the Human Performance attribute of the Mitigating Systems Cornerstone, in that simulator deviations could lead to pre- and post-event human error. The finding is of very low safety significance since the finding is only related to simulator fidelity. (Section 4OA5).

B. Licensee Identified Violations

One violation of very low significance, which was identified by the licensee, was reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and corrective action are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status:

Unit 1 operated essentially at 100 percent power throughout the inspection period.

Unit 2 operated essentially at 100 percent power throughout the inspection period, with a number of exceptions. In July and August the unit down-powered less than 5 percent power on five occasions due to degrading main condenser vacuum conditions as a result of external conditions. Also, between 7/9-11/04, the unit down-powered to approximately 75 percent power to perform main condenser tube leak identification and repair activities. In September (9/9 and 9/24), the unit was down-powered to support calibration and surveillance activities. In addition, between 9/24-25/04, the unit was down-powered to 94 percent power to effect repairs on a feed system valve actuator.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01 - 1 sample)

a. Inspection Scope

The inspectors reviewed licensee actions due to the effects of Hurricane Ivan between September 17-22, 2004. Specifically, the inspectors reviewed actions taken following the entry into Abnormal Operating Procedure (AOP) 1/2OM-53C.4A.75.2, "Acts of Nature - Flood," Rev. 20, when the Ohio River level exceeded a level of 670 feet mean sea level (MSL). The inspectors performed various plant walkdowns, reviewed operator logs and interviewed plant personnel to evaluate FENOC's execution of the flood AOP.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04)

a. Inspection Scope

Partial System Walkdowns (3 samples)

The inspectors performed three partial system walkdowns during this inspection period. The inspectors evaluated the operability of a selected train or system while the redundant train or system was inoperable or unavailable. The inspectors verified correct valve positions and breaker alignments in accordance with applicable procedures, and verified consistency with applicable chapters of the Updated Final Safety Analysis Report. The following walkdowns were performed:

- On August 18, 2004, the inspectors performed a walkdown of the Unit 2 No. 1 emergency diesel generator (EDG) while the No. 2 EDG was out of service for planned maintenance.

- On September 9, 2004, the inspectors performed a walkdown of the Unit 1 ‘A’ train Recirculation Spray (RS) system while the ‘B’ outside RS pump was out of service for motor preventive maintenance.

- On September 28, 2004, the inspectors performed a walkdown of the standby service water system in the auxiliary intake structure while the Unit 2 ‘A’ service water pump and ‘D’ intake bay were out of service for planned maintenance.

Complete System Walkdown (1 sample)

The inspectors conducted a detailed review of the alignment and condition of the Unit 1 Main Steam (MS) System. This system was selected based on its risk significance and the results of previous inspections. The inspectors reviewed plant drawings, abnormal operating procedures, and emergency operating procedures to determine proper equipment alignment. The inspectors reviewed and evaluated the impact on the MS system operation due to work orders based on existing deficiencies. Condition reports associated with the MS system were also reviewed to verify that the licensee was adequately identifying and correcting system deficiencies. In addition, the inspectors performed a detailed review of the MS system health report and the design basis document in order to gain insights on any longstanding issues.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05 - 9 samples)

a. Inspection Scope

The inspectors reviewed the Unit 1 Updated Fire Protection Appendix ‘R’ Review, Rev. 25, and the Unit 2 Fire Protection Safe Shutdown Report, Addendum 27, and identified the risk significant areas listed below for inspection. The inspectors also reviewed the fire protection attributes of the listed fire areas to verify compliance with the documents described above, and the criteria delineated in Administrative Procedure 1/2-ADM-1900, “Fire Protection,” Rev. 9. This review included FENOC's control of transient combustibles, material condition of fire protection equipment utilized for suppression and detection, condition of fire barriers and rated doors, and the adequacy of any fire protection compensatory measures, as applicable. In addition, the inspectors evaluated the adequacy and scope of the pre-fire plans established for the areas.

- Unit 1 West Cable Vault (Fire Area CV-1)
- Unit 1 East Cable Vault (Fire Area CV-2)
- Unit 1 Pipe Tunnel Area (Fire Area PT-1)
- Unit 2 Alternate Shutdown Panel Room (Fire Area ASP)
- Unit 2 Control Building Computer Room (Fire Area CB-4)
- Unit 2 Control Building West Communication Room (Fire Area CB-6)
- Unit 2 Cable Vault and Rod Control Area (Fire Area CV-5)

- Unit 2 Cable Vault and Rod Control Area Relay Room (Fire Area CV-6)
- Unit 2 Fuel Building (Fire Area FB-1)

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06 - 1 sample)

a. Inspection Scope

The inspectors reviewed the Unit 1 Updated Final Safety Analysis Report (UFSAR) and the Individual Plant Examination (IPE) to evaluate the design basis and risk significance for internal floods. Based on associated risk significance, the inspectors performed walkdowns between August 2 - 6, 2004, of the plant areas listed below. During these walkdowns, the inspectors examined a sample of internal flood seals, inspected the material condition of potential sources of internal flooding, and verified the performance of various floor drains, sump pumps, and level alarm circuits.

- CR-2 (Control Room HVAC Equipment Room)
- CR-3 (Communication Equipment and Relay Panel Room)
- CR-4 (Process Instrument and Rod Position Room)
- NS-1 (Normal Switchgear Room)
- CS-1 (Cable Spreading Room)

b. Findings

Inadequate Test Control Associated With Unit 1 Flood Control Level Switches

Introduction. The inspectors identified a NCV for failure to properly implement test control measures to ensure the operability of Unit 1 flood control level alarms. The finding was of very low safety significance because operator rounds each shift would promptly alert the control room personnel of internal or external flooding conditions.

Description. During the walkdowns of the selected flood control areas, the inspectors noted various curbs and level switches. These curbs and switches are described in the UFSAR, as well as the IPE, and function to collect leakage through wall penetrations which are located below the probable maximum flood elevation. A level alarm switch, consisting of a mechanical float with associated electronics, is provided in each curbed area in the selected flood areas, with the exception of area CR-3. The inspectors noted that a penetration described in the UFSAR between CR-2 and CR-3 had been plugged, and thus the level switch in area CR-2 would not alarm if a high level occurred in area CR-3. This deviation was brought to the attention of the system engineer and documented in a condition report.

The inspectors requested to review the testing documentation for the four associated level switches, LS-1DA-117, -118, -119, and -120. However, no evidence could be located which indicated that these switches were ever tested since initial construction of the plant. FENOC subsequently tested all four switches by mechanical actuation; they were noted to move freely and actuated a common alarm annunciator in the control room. However, when tested to simulate a flooding condition using water, all four switches failed to actuate the alarm. Further investigation revealed that the switches had a solid cylindrical housing installed around the float that was not properly vented. As a result, entrapped air prevented the float from rising and actuating in the event of a flood. Holes were ultimately drilled in the housing to provide a vent path; the four level switches successfully actuated the control room alarm during the post maintenance testing.

Analysis. The issue was more than minor because it was associated with the protection against external factors performance attribute of the mitigating systems cornerstones. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors conducted a significance determination process (SDP) Phase 1 screening and determined that the finding involved the loss of a safety function, identified by the licensee through an IPE, that contributes to external event initiated core damage accident sequences. Due to the external events initiator, a Phase 3 evaluation was performed in lieu of a Phase 2 SDP evaluation.

Level switches LS-1DA-118, 199 and 120 were installed to detect external flooding through exterior wall penetrations. This event was considered a slowly evolving scenario such that operators could take mitigating actions to prevent equipment damage. Thus these level switches had minimal risk significance.

The purpose of the fourth level switch, LS-1DA-117, is to alert the CR operators to an internal flooding condition in the control room (CR) ventilation room. The most likely cause of flooding would be the failure of the river water piping that supplies cooling to the CR ventilation equipment. A flooding scenario that results in six inches of water in the adjacent process area is assumed to cause a plant trip due to shorting in the relay racks. In addition to causing a reactor trip, the flooding can result in damage to the solid state protection system and complicate the transient. The analysis estimated that the probability of a river water pipe break to be in the range of low E-3 per reactor year. Also, assuming that the level switches were functioning, the probability that the operators would fail to isolate the flooding prior to the water level reaching six inches in the process rack area was estimated to be in the range of low E-3. Therefore, the frequency of a flooding event, initiated by a river water pipe break in the CR vent room, that results in a reactor a trip would then be estimated to be low E-6 per reactor year.

The control room flood core damage frequency in the BVPS Unit 1 PRA is low E-9 per reactor year. By using this value and the above calculated initiating event frequency, a conditional core damage probability (CCDP) can be calculated. The CCDP is determined by dividing the CDF by the initiating event frequency. This would result in a CCDP in the mid E-4 range.

In the condition identified by the inspectors, in which the level switch was inoperable, the only remaining flood detection would come from plant personnel tours in the area, or by CR indications that would alarm on low river water pressure or low flow. Given the loss of the level switch function, it was estimated that there would be a factor of 100 increase in the probability that the operators would fail to isolate the flooding prior to the level reaching six inches in the process rack area. This would result in the frequency of a flooding event, initiated by a river water pipe break in the CR vent room, that results in a reactor trip to be in the low E-4 per reactor year range.

The CDF for the flooding scenario, in which the level switches failed to function, would be the product of the above initiating event frequency times the CCDP. This product would result in CDF in the low E-7 per reactor year range.

The increase in CDF ()CDF) was estimated to be in the low E-7 range. This is calculated by taking CDF for the condition in which the level switches would not function, and subtracting the baseline CDF. Given a)CDF in this range, using MC 0609 Appendix H, the potential for an increase in the large early release frequency ()LERF) was negligible, because the Beaver Valley Unit 1 containment is of the subatmospheric design. Based on this comprehensive evaluation of the initiation event frequency, surviving mitigating systems and operator actions to mitigate the impact of the flooding event, the finding was considered to have a very low safety significance (Green).

This finding was related to the Problem Identification and Resolution cross cutting area in that FENOC failed to identify the lack of adequate testing of these flood switches since initial construction.

Enforcement. 10 CFR 50, Appendix B, Criterion XI "Test Control," requires, in part, that all testing required to demonstrate that structures, systems, and components will perform satisfactorily is identified and performed in accordance with written test procedures. Contrary to these requirements, four flood level switches were identified that had never undergone appropriate testing to verify the switches were operational. Because this test control deficiency was of very low safety significance and has been entered into the corrective action program as CR 04-06173, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000334/2004005-01, Inadequate Design Control Associated With Unit 1 Flood Control Level Switches.

1R11 Licensed Operator Requalification Program (71111.11)

1. Resident Inspector Quarterly Review of Requalification Training (71111.11Q - 1 sample)

a. Inspection Scope

The inspectors observed the conduct of Unit 2 licensed operator requalification training evaluations conducted in the plant reference simulator on September 21, 2004. The inspectors observed licensed operator performance, with a particular focus on effective communications, implementation of abnormal and emergency operating procedures, command and control, technical specification compliance, and emergency plan implementation (See Section 1EP6). The inspectors evaluated simulator fidelity to verify major plant configurations or changes were captured in the simulator to ensure adequate training was provided. Inspectors evaluated the staff evaluators during the examination to ensure deficiencies in operator performance were properly identified, and that identified conditions adverse to quality were appropriately entered into the corrective action program for resolution. Other documents utilized in this inspection include the following:

- 1/2-ADM-1351, Rev. 2 Licensed Operator Retraining Program
- 1/2-ADM-1357, Rev. 5 Conduct of Simulator Training
- 1/2-ADM-1359, Rev. 7 Simulator Configuration Control

b. Findings

No findings of significance were identified.

2. Biennial Licensed Operator Requalification Program Inspection (71111.11 - 1 sample)

a. Inspection Scope

The following inspection activities were performed using NUREG-1021, Draft Revision 9, “Operator Licensing Examination Standards for Power Reactors,” Inspection Procedure Attachment 71111.11, “Licensed Operator Requalification Program,” and 10 CFR 55.46, ”Simulator Rule,” as acceptance criteria.

- The inspectors reviewed documentation of operating history since the last requalification program inspection, which included discussions of facility operating events with the resident staff. Documents reviewed included NRC inspection reports and condition reports 02-09729, 03-06962, 03-08458, and 04-04063, which involved human performance issues. The purpose of this review was to determine whether any plant events were indicative of training deficiencies.

- The inspectors reviewed three Unit 1 2004 comprehensive written exams, and one Unit 1 2004 annual operating test (consisting of two scenarios and five job performance measures). The inspectors also observed the administration of the annual operating test to one staff crew and one on-shift crew. The purpose of these reviews and assessments was to determine whether exam quality and exam administration met the criteria of the Examination Standards and 10 CFR 55.59.

- The inspectors interviewed one instructor, the licensed operator requalification (LOR) program administrator, two reactor operators (RO), and two senior reactor operators (SRO) for feedback regarding the implementation of the Unit 1 LOR program to determine whether training staff modified the program when appropriate. In addition, five plant and industry events or changes were reviewed to verify that these items were adequately addressed in the LOR program.

- Inspectors reviewed remedial training packages for two individuals that failed evaluations during the current two-year (January 1, 2003 to December 31, 2004) cycle.

- Inspectors reviewed the following records to verify operators were complying with license conditions:

  - A sample of Unit 1 attendance records (six records) for the current two-year training cycle

  - A sample of medical records (five from Unit 1; five from Unit 2). Records were checked to verify that 1) restrictions noted by the doctor were reflected on the individual's license; and 2) the physical exams were given within the last 24 months

  - A sample of Unit 1 license renewals (six records), proficiency watch-standing (three records), and license reactivations (two records)

- The inspectors observed Unit 1 simulator performance during the conduct of the examinations and reviewed Unit 1 simulator performance tests and discrepancy reports to verify compliance with the requirements of 10 CFR 55.46.

b. Findings

Inspectors identified an issue related to the development of the Unit 1 LOR comprehensive written exam. When choosing what subjects were to be tested, the training staff sampled subjects taught during the latter part of the previous LOR period and most of the current LOR period, but purposely excluded subjects taught during the latter part of the current LOR period. Inspectors noted that this method of testing was consistent with 1/2-ADM-1351, "Licensed Operator Retraining Program," which uses a systems approach to training (SAT) methodology in lieu of paragraphs (c)(2), (3), and (4) of10 CFR 55.59. Inspectors noted that this method of testing may not meet the intent of the "in lieu of" section of 10 CFR 55.59 (c). 10 CFR 55.59 (c)(4), part (ii) states, in part that the LOR program must include "Written examinations which determine ... knowledge of subjects covered in the requalification program ..." Although it is acceptable for the written exam to test knowledge of subjects covered during previous LOR programs, the inspectors could not determine whether it was also acceptable for the written exam to exclude subjects taught during the current LOR program, namely the topics yet to be taught in the remainder of the period through December 2004. As a result, this issue is unresolved pending further review by NRC staff. (URI 05000334, 412/2004005-02, Acceptability of licensee's LOR written exam development methodology.)

1R12 Maintenance Rule Implementation (71111.12 - 3 samples)

a. Inspection Scope

The inspectors evaluated Maintenance Rule (MR) implementation for the issues listed below. The inspector evaluated specific attributes, such as MR scoping, characterization from the system basis documents of failed SSCs, MR risk categorization of SSCs, SSC performance criteria or goals, and appropriateness of corrective actions. The inspectors verified that the issues were addressed as required by 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance of Nuclear Power Plants,” and 1/2-ADM-2114, “Maintenance Rule Program Administration,” Revision 1. For selected systems, the inspectors evaluated whether system performance was properly dispositioned for MR category (a)(1) or (a)(2) performance monitoring, and reviewed the system basis documents. The following conditions were evaluated:

- CR-04-06393 Failure of FT-FW-487 Results in a Unit 1 Secondary Calorimetric Power Transient
- CR 04-05557 Vital Bus 2-1 Loss of Inverter
- CR 04-05619 2SWS-P21A Discharge Check Valve (2SWS-57) Found Partially Closed During PM

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Control (71111.13 - 7 samples)

a. Inspection Scope

The inspectors reviewed the scheduling and control of seven activities, and evaluated the effect on overall plant risk. This review was against criteria contained in documents located in the Attachment. The inspectors reviewed the planned or emergent work for the following activities:

- The inspectors reviewed the risk assessment associated with the unexpected failure of the emergency response facility (ERF) diesel generator on July 28, 2004. This emergent risk assessment captured the increased risk associated with the unexpected failure, and included periods of unavailability associated with concurrent emergency diesel generator (EDG) maintenance and surveillance activities. See Section 4OA7 for further details.

- The inspectors evaluated the risk assessment of a planned activity involving a 480V breaker replacement associated with Motor Operated Valve (MOV)-RW-113D1, on August 9, 2004. This MOV is one of two valves that provides cooling water to the Unit 1 No. 2 EDG.

- The inspectors reviewed an emergent work activity associated with the Unit 1 No. 1 EDG from August 23 to August 25, 2004, following a planned maintenance outage and subsequent failure of the No. 1 EDG to start (see section 1R15). The inspectors reviewed the risk assessment associated with this emergent activity, compliance with technical specifications, and associated corrective actions.

- The inspectors reviewed the risk assessment associated with an emergent circuit card failure associated with the turbine-driven auxiliary feedwater pump, and subsequent repairs that occurred between September 10-11, 2004.

- On September 16, 2004, the inspectors reviewed the risk assessment associated with an uncoupled run of the Unit 1 ‘C’ River Water (RW) pump. This activity rendered the ‘B’ train of RW out of service and unavailable since only one RW pump breaker can be racked onto the 4160 volt switchgear at a time. This planned activity increased the maintenance risk threshold from green (<2 times baseline) to yellow (2 to 10 times baseline core damage frequency).

- On September 23, 2004, the inspectors reviewed the risk assessment associated with the surveillance of the ‘A’ train of the Unit 1 solid state protection system. This surveillance was conducted in accordance with 1MSP-01.04, “Solid State Protection System Train ‘A’ Bi-Monthly Test,” Rev. 22. Issues noted during the performance of this surveillance were documented in CR 04-07201. This planned activity increased the maintenance risk threshold from green (<2 times baseline) to yellow (2 to 10 times baseline core damage frequency).

- On September 28, 2004, the inspectors reviewed the risk assessment associated with a planned maintenance activity on the Unit 2 ‘C’ recirculation spray pump. This activity involved the replacement of 4160V cell switch and rendered the pump out of service and unavailable. The inspectors also reviewed work order (WO) 200092127 and discussed the activity with the maintenance engineer.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions (71111.14 - 3 samples)

1. Non-routine Plant Evolutions

a. Inspection Scope

The inspectors reviewed human performance during the following non-routine plant evolution, to determine whether personnel performance caused unnecessary plant risk or challenges to reactor safety. The inspectors evaluated whether the evolution was properly implemented according to the applicable procedures and Technical Specification (TS) limiting condition for operations (LCOs).

- On August 14, 2004, during the performance of 1-Operations Surveillance Test (OST)-1.1, “Control Rod Assembly Partial Movement Test,” Rev. 11, the operators noted a position deviation of the steps between groups 1 and 2 while inserting shutdown bank ‘A’. Initial investigations did not reveal an obvious cause, and the OST was terminated. Operators entered TS LCO 3.1.3.5, since the shutdown bank ‘A’ was not withdrawn at or above its core operating limit report (COLR) limit of 225 steps, and subsequently, TS LCO 3.0.3, since more than one shutdown rod was still inserted beyond the COLR limit in excess of one hour. While initial troubleshooting determined that the position indication anomaly was a failure of the shutdown bank group 2 demand indication, further troubleshooting indicated that the rod control system was failing to generate an inward demand signal. As a result, FENOC entered TS LCO 3.1.3.1.d, which requires the control rods to be restored to operable status within 72 hours. On August 19, circuit cards were replaced and the rod control system was returned to service following successful completion of 1OST-1.1.

- The inspectors evaluated the licensee’s response to an unexpected, automatic start of the Unit 1 turbine-driven auxiliary feedwater pump on September 10, 2004. The cause was later determined to be a failed circuit card located in the ‘B’ Train of Solid State Protection System, which was replaced and returned to service on September 11, 2004. The inspectors reviewed shift narrative logs, technical specifications (for compliance and operability concerns), and alarm response procedures, to verify appropriate actions were taken. The inspector assessed the adequacy of FENOC's interim corrective actions and verified this event was entered into the corrective action program for resolution as 04-06918.

b. Findings

No findings of significance were identified.

2. (Closed) Licensee Event Report (LER) 05000334/2003007-00: Inadvertent Reactor Trip During Solid State Protection System Testing

This event was discussed in NRC Inspection Report No. 05000334,412/2003005. No new issues were revealed by the LER. This LER is closed.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 6 samples)

a. Inspection Scope

The inspectors reviewed the following six conditions to determine whether proper operability determinations (OD), Basis For Continued Operations (BCO), or applicable assessments were performed. In addition, where applicable, the inspectors verified that Technical Specification (TS) limiting conditions for operation (LCO) requirements were properly addressed.

- The inspectors reviewed an OD associated with the Unit 1 No. 1 Emergency Diesel Generator (EDG), as documented in CR 04-06592, regarding the failure of the EDG to start on August 25, 2004. The inspectors assessed the adequacy and acceptability of FENOC's conclusion in the OD that the 1-1 EDG would have started and powered the AE emergency 4160V bus, given a valid start signal on undervoltage or a safety injection actuation. The inspectors evaluated the contingency actions, which included maintaining the start circuit No. 2 selected until start circuit No. 1 was returned to service.

- The inspectors reviewed an engineering evaluation regarding the operability aspects associated with scaffolding attached to the motor of the Unit 2 ‘C’ SW pump. The inspectors held discussions with design engineers regarding the seismic qualification of the pump, as well as the engineering assessment of past operability. The inspectors also reviewed FENOC’s conclusion that the attached scaffold would have had an insignificant effect on the pump’s safety function had a seismic event occurred during pump operation.

- The inspectors reviewed the licensee’s operability assessment of the Unit 1 turbine-driven auxiliary feed water (TDAFW) pump, due to excessive vibration of the pump suction flow indication, Flow Indicating Switch (FIS)-1FW-152, and the apparent cycling of an associated solenoid valve. The inspectors reviewed CR-40-06162 and the associated OD, and evaluated the adequacy of the licensee’s conclusion that the TDAFW pump was unaffected by the vibration.

- The inspectors reviewed BCO 1-04-003, which dealt with a discrepancy involving the amount of containment metal mass between the original design data for Unit 1 and the current atmospheric containment conversion (ACC) project. The calculation for the ACC concluded that a total mass of 5.7 million pounds exists while the original calculations concluded 4.2 million pounds. The metal mass is used as a heat sink input to the design basis accident (DBA) loss of coolant accident (LOCA) containment and reactor core analysis of record. This BCO provided an interim operability assessment for the current core burnup until a detailed reanalysis using the larger containment metal mass could be performed. The justification for continued plant operation included various conservatisms which showed that peak cladding temperatures would remain below 2200 degrees during a DBA LOCA as prescribed by 10CFR 50.46, and peak containment pressure would remain within analyzed limits.

- The inspectors evaluated the licensee's basis for continued operation of the Unit 1 'A' Reactor Coolant Pump due to a leak in the seal injection line (BCO1-04-001). The inspectors evaluated the licensee's conclusion that the pump and seal remained operable, based primarily on the ability of the thermal barrier heat exchanger to reduce the heat from the reactor coolant to avoid damage to the lower radial bearing in the event of a loss of seal injection flow.

- The inspectors reviewed an OD associated with a potential non-conformance that involved the Unit 1 atmospheric dump valves (ADVs), and whether the ADVs were consistent with the current licensing basis as documented in CR 04-06293. The inspectors assessed the adequacy and acceptability of FENOC's conclusion in the OD, specifically, that the ADVs would continue to perform their design basis function by assisting operators during the plant cooldown following the design basis steam generator tube rupture, and assuming the worst case single failure.

b. Findings

Inadequate Procedural Adherence During the Installation of Scaffolding

Introduction. A Green NCV was identified for failure to correctly implement a scaffolding procedure in accordance with the requirements of TS 6.8.1.a.

Description. On July 19, 2004, while performing a walkdown of the intake structure, the inspectors noted a scaffold brace bar attached to the motor lifting lug of the Unit 2 'C' safety-related service water (SW) pump. This potential non-conforming condition was brought to the attention of the shift manager, and along with the building services supervisor, verified that the scaffold was not erected in accordance with procedure 1/2-ADM-0810, "Scaffold Erection and Tagging," Rev. 4. This procedure precludes scaffold from being in contact with, supported by, or braced to safety-related or seismically qualified equipment, and was immediately removed. The scaffold was originally erected on July 9 in order to support maintenance associated with the motor coil cleaning of the 'C' SW pump motor (see Section 1R19). At the time of discovery on July 19, the pump was considered the spare pump and not required for TS compliance. However, from July 9 to July 12, and from July 15 to July 16, the pump was in service and considered operable per TS 3.7.4.1. An engineering assessment was performed to verify past operability and concluded that the additional weight of the scaffold was insignificant unless a seismic event occurred. Given that, the analysis calculated that the motor would experience an additional 525 pounds of dynamic force during a design basis earthquake. This force was determined to be well within the motor stand capability. Thus the pump remained operable even during an earthquake.

Analysis. The finding adversely impacted the Unit 2 ‘C’ SW pump reliability and capability. Because the finding affected the reactor safety mitigating system cornerstone objective, the finding is greater than minor. Utilizing, Appendix A of IMC 0609, “Significance Determination Process,” the inspectors performed a Phase 1 analysis. This finding was determined to be of very low safety significance, Green, since the deficiency was determined to not result in a loss of function per Generic Letter 91-18 as documented by the engineering seismic assessment.

This finding involved the cross-cutting area of human performance, due to the failure to properly implement a scaffold procedure that prohibited the attachment of scaffold to safety-related components.

Enforcement. TS 6.8.1 requires written procedures be established, implemented, and maintained as recommended by Appendix ‘A’ of Regulatory Guide 1.33, Rev. 2. This appendix lists various examples of procedures used at commercial nuclear facilities, including procedures which implement maintenance which can affect the performance of safety-related equipment. Contrary to the above, FENOC failed to correctly implement a scaffold erection procedure which adversely affected a safety-related SW pump. Because this adverse condition was of very low safety significance and has been entered into the corrective action program as CR 04-05739, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000412/2004005-03, Inadequate Procedural Adherence During the Installation of Scaffolding.

1R16 Operator Work-Arounds (Cumulative Review) (71111.16 - 1 Sample)

a. Inspection Scope

The inspectors reviewed the current listing of active Operator Work-Arounds (OWAs) for Beaver Valley 2, which also included Operator Challenges and Control Room Deficiencies. The review was conducted to verify that the cumulative effects of known OWAs were evaluated to determine the overall impact on the affected systems. While the current listing contained zero OWAs, the inspectors assessed the cumulative impact of overall deficiencies and challenges to control room operators to determine if it adversely affected the ability of plant operators to implement emergency procedures or respond to plant transients. The inspectors reviewed the deficiencies and challenges and verified that they were being captured for resolution, and reviewed the guidance contained in BVBP-OPS-0002, Rev. 9, “Operator Work Arounds, Operator Challenges, and Control Room Deficiencies.”

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17A - 1 sample)

a. Inspection Scope

Annual. The inspectors reviewed one permanent plant modification, Engineering Change Package (ECP) 04-0068, “Removal of Spool Piece Connecting Suction Piping Vent to the Volume Control Tank (VCT).” This ECP removed vent piping that connected the suction piping of the Unit 2 ‘A’ and ‘C’ charging pumps to the VCT. CR 04-00980 described a condition where a void was detected in the suction piping of the ‘C’ charging pump. Removal of this vent piping, along with the installation of a different style capped vent valve, was designed to eliminate a potential backflow path to the charging pump suction from the VCT, and prevent hydrogen accumulation in the spare pump. The inspectors verified that the existing design bases, licensing bases, and performance capability of the charging system was not degraded by this modification.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19 - 6 samples)

a. Inspection Scope

The inspectors reviewed and/or observed six post-maintenance tests (PMTs) to ensure: 1) the PMT was appropriate for the scope of the maintenance work completed; 2) the acceptance criteria were clear and demonstrated operability of the component; and 3) the PMT was performed in accordance with applicable procedures. The following PMTs were observed:

- 1OST-1.10, “Cold Shutdown Valve Exercise Test,” Rev. 28, performed on August 16, 2004, following the replacement of the breaker associated with valve, 1MOV-MS-105.

- 1OST-7.6, “Centrifugal Charging Pump [1CHS*P11C],” Rev. 23, performed on June 19, 2003, following bearing inspection and seal replacement.

- 1OST-47.3E, “Containment Isolation and ASME Section XI Test - Work Week 1,” Rev. 3, performed on August 10, 2004, following the molded case circuit breaker replacement associated with MOV-1CH-115D, “Refueling Water Storage Tank Discharge to Charging Pumps Suction Valve.”

- 2OST-30.6B, “Service Water Pump [2SWS*P21C] Test on Train ‘B’ Header,” Rev 11, performed on July 15, following motor cleaning.

- 2OST-7.5, “Centrifugal Charging Pump [2CHS*P21B],” Rev. 27, performed on July 16, 2004, following the performance of preventative maintenance in accordance with 1/2PMP-7CH-P-1A/21A-B-C-1M, “Charging/High-Head Safety Injection Pump Lubrication and Maintenance,” Rev. 13.

- 2OST-30.2, “Service Water Pump [2SWS*P21A] Test,” Rev. 27, performed on August 5, 2004, following replacement of the pump discharge check valve, 2SWS-57.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors evaluated selected receipt, movement and inspection activities associated with new fuel assemblies in preparation for the upcoming Unit 1 refueling outage. Specifically, the inspectors verified activities were performed in accordance with OM-16, “Site Receipt and Handling of New Fuel Assemblies and Shipping Containers,” Revision 7. In addition, the inspectors verified that appropriate fuel movement accountability was maintained, and that identified adverse conditions were entered into the corrective action program for resolution.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22 - 6 samples)

a. Inspection Scope

The inspectors observed and/or reviewed the following six surveillance tests. This review verified that the equipment or systems were capable of performing their intended safety functions and to ensure compliance with related technical specifications, updated final safety analysis report, and associated procedural requirements:

•	1OST-49.1, Rev. 8	Shutdown Margin Calculation (Plant Critical)
•	1OST-24.4, Rev. 28	Steam Turbine Drive Auxiliary Feed Pump Test [1FW-P-2]
•	2OST-36.2, Rev. 42	Emergency Diesel Generator [2EGS*EG2-2] Monthly Test
•	2OST-47.1, Rev. 8	Containment Air Lock Test
•	2OM-54.4.C1, Rev. 13	Daily Heat Balance, conducted on September 14, 2004.
•	1OST-16.2, Rev.7	Supplementary Leak Collection And Release Test For Exhaust Through The Main Filter Bank - Train B

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06 - 1 sample)

Simulator-Based Evaluation

a. Inspection Scope

The inspectors observed a Unit 2 operator requalification simulator evaluation (See Section 1R11) and evaluated operator performance regarding event classifications. The simulator evaluation involved multiple safety-related component failures and plant conditions that warranted a simulated Alert emergency event declaration. The licensee counted this evolution toward Emergency Preparedness Drill/Exercise Performance (DEP) Indicators, therefore the inspectors reviewed the classifications to determine whether they were appropriately credited. Additionally, the inspectors verified the DEP performance indicators were properly evaluated consistent with Nuclear Energy Institute (NEI) 99-02, Rev. 2, "Regulatory Assessment Performance Indicator Guideline." Other documents utilized in this inspection include the following:

- 1/2-ADM-1111, Rev. 2 — NRC EPP Performance Indicator Instructions
- EPP/I-1a, Rev. 7 — Recognition and Classification of Emergency Conditions
- EPP-I-3, Rev. 19 — Alert
- EPP-I-4, Rev. 7 — Site Area Emergency

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification (71151 - 2 samples)

Safety System Functional Failures

a. Inspection Scope

The inspectors reviewed the Unit 1 and Unit 2 performance indicators for safety system functional failures to determine whether the NRC-approved guidance, provided in NEI 99-02, was properly implemented. Verification included review of the data collected, PI definitions, data reporting elements, calculational methods, definition of terms, and use of clarifying notes. The inspectors verified accuracy of the reported data through reviews of LERs submitted during the period August 2003 through August 2004.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution

1. Annual Sample Review (1 sample)

CR-03-11082 - 2RSS-P21A Breaker Closure While Racking

a. Inspection Scope

The inspectors selected CR 03-11082 for detailed review. The CR was initiated in October 2003 and documented a spurious closing and reopening of the circuit breaker for the 2RSS-P21A pump. A similar event occurred in February 2002, and was documented in CR 02-01431. These events occurred during installation (racking in) of the breaker into the switchgear. The inspector reviewed the adequacy and appropriateness of FENOC's actions to address these events. Specifically, the inspector reviewed actions taken to assess operability of the system following each event, root cause evaluations and corrective actions.

The following documents were also reviewed during this inspection:

- 1/2-PMP-E-36-015 ITE Medium Voltage Circuit Breaker Inspection and Test Model 5HK-250/350, Rev. 11
- Vendor Manual IB 6.2.7D Installation/Maintenance Instructions - Medium Voltage Power Circuit Breakers, Type 5HK 1200 thru 3000 Amperes 5000 Volts
- 12241-E-5DQ Elementary Diagram 4160 Volt, Recirc Spray Pump (2RSS*P21A)
- Waukesha Electric Systems, Inc. Failure Analysis Report for Breaker Serial Number 51958B-120036, dated June 4, 2004

b. Findings and Observations

No findings of significance were identified. The inspector noted that testing and troubleshooting actions following the first event were appropriate. The problem could not be repeated and the circuit breaker was returned to service following the performance of inspections, preventive maintenance checks and surveillance tests. No deficiencies were identified during these activities.

Following the second failure, the system was returned to service primarily based on actions taken after the first event, satisfactory system performance during quarterly testing performed in the period between the two events, and satisfactory breaker operation during a test following the second event. The second failure caused plant personnel to suspect that the cause was related to the removal and reinstallation of the circuit breaker in the switchgear. However, no additional troubleshooting was performed to explore this area further.

The inspector found that FENOC's conclusions that the system was operable were correct. The conclusions were also supported by the vendor root cause determination that was performed following the breaker replacement. However, the inspector noted that the failure to perform additional troubleshooting (focused on breaker removal/reinstallation) immediately following the second event was a missed opportunity for earlier identification of the root cause and an opportunity to establish a more conclusive basis for continued operability.

2. Inspection Module Problem Identification and Resolution (PI&R) Review

a. Inspection Scope

The inspectors reviewed various CRs associated with the inspection activities captured in each inspection module detailed in this report. During this review, the inspectors assessed the fundamental ability of the licensee to identify adverse conditions for the areas inspected, and verified the licensee had entered these issues into its corrective action program for resolution. Where applicable, CRs reviewed during the inspection are documented under each module; however, for reviews that entailed large number of CRs, these are more appropriately documented in the Attachment.

b. Findings

No findings of significance were identified.

3. Daily Condition Report Review

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. This review was accomplished by reviewing hard copies of each condition report, attending various daily screening meetings, and when necessary, by accessing the licensee's computerized corrective action program database.

b. Findings

No findings of significance were identified.

4. Cross-References to PI&R Findings Documented Elsewhere

Section 1R06 describes a finding for failure to identify a condition adverse to quality which challenged safety-related equipment in the event of an internal or external flood. Consequently, since initial construction, advanced warning of certain flooding conditions has never been available to the control room operators.

4OA3 Event Follow-up (71153 - 1 sample)

(Closed) LER 05000412/2003003-00: Automatic Reactor Trip Due to Low Steam Generator Water Level

This event was discussed in NRC Inspection Report No. 50-334/412/03-05. No new issues were revealed by the LER. This LER is closed.

4OA4 Cross Cutting Aspects of Findings

Section 1R15 describes a finding for failure to failure to implement a scaffolding procedure. Consequently, a scaffold was attached to a safety-related SW pump, thus impacting the pump's reliability. This finding exhibited human performance cross cutting aspects because the procedure clearly prohibited the installation practice used.

4OA5 Other

(Closed) URI 05000334,412/2003-005-01: Acceptability of licensee's simulator testing methodology.

Introduction. While evaluating simulator testing during the November 2003 Unit 2 LOR program inspection, inspectors identified that the licensee compared current year transient test data to prior year test data rather than to "best estimate" data as specified in ANSI/ANS 3.5-1985; the facility tested both simulators in this manner. Additional details from that inspection are in NRC Inspection Report 05000334,412/2003005.

Description. Because the licensee's methodology deviated from ANSI guidance for simulator testing, inspectors noted the potential existed for deviations to be introduced between the plant control rooms and the plant reference simulators. Deviations could cause negative training, which in turn could have an adverse effect on operator actions during plant operations. In response to this issue, the licensee initiated corrective actions to ensure their simulator testing methodology conformed to ANSI guidance. As a result of their actions, the training staff identified four examples where simulator response deviated from plant response. These deviations were documented in simulator deficiency reports (SDRs): SDR-6027, SDR-6028, SDR-6029, and SDR-6030. The inspectors determined from a review of these deviations that they could result in negative training of operators.

Analysis. This finding was more than minor because it affected the Human Performance attribute of the Mitigating Systems Cornerstone, in that simulator deviations could lead to pre- and post-event human error. Inspectors referred to the Operator Requalification Human Performance SDP (Appendix I) and used the questions associated with Flowchart Block #12, “Could deviations or the differences between the plant ... and the ... simulator negatively impact operator actions? ... Could the differences result in negative training?”, Based on a “Yes” answer, the inspectors determined the Finding was of very low safety significance (Green).

Enforcement. Although the facility did not conduct testing in accordance with the applicable ANSI standard, compliance with that standard is not a regulatory requirement, and therefore, no violation of regulatory requirements occurred. The facility entered the Finding into their corrective action program as Condition Report No. 04-00994. The licensee also modified their testing methodology to conform with the applicable ANSI requirements. The URI is closed. FIN 05000334, 412/2004005-04, Acceptability of Licensee’s Simulator Testing Methodology.

4OA6 Meetings, Including Exit

The inspectors presented inspection results to the licensee at the conclusion of the inspection on October 25, 2004. No materials reviewed were identified by the licensee as proprietary.

4OA7 Licensee-Identified Violations

The following violations are of very low safety significance, were identified by the licensee and are violations of NRC requirements which meet the (Green) criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCVs:

10 CFR 50.65(a)(4), requires in part, that before performing maintenance activities, the licensee shall assess the increase in risk that may result from proposed maintenance activities. Contrary to this requirement, FENOC failed to adequately assess the resultant risk from concurrent maintenance and surveillance activities on the Unit 1 emergency diesel generator (EDG) and the emergency response facility diesel generator (ERF DG). Specifically, on July 27, 2004, FENOC inappropriately determined that the ERF DG was available (relative to the maintenance rule) following the completion of maintenance, and prior to the completion of a successful post-maintenance test (PMT). Subsequently on July 28, following the performance of a planned Unit 1 EDG surveillance activity, the PMT performed for ERF DG maintenance the previous day was unsuccessful due to a component failure associated with the ERF DG cooling system. As a result, FENOC did not perform a risk assessment that captured the concurrent unavailability periods of the ERF DG due to the failure, and the Unit 1 EDG during surveillance testing. The failure to adequately assess the increase in risk while the ERF DG was functionally unavailable, including the time following the PMT failure, is considered a violation of very low safety significance and is being treated as an NCV. The safety significance was evaluated using the Appendix A of the NRC's Significant Determination Process in IMC-0609, and

was determined to be Green. This event was entered into FENOC's corrective action system as CR 04-06012.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

R. Boyle	System Engineer
S. Checketts	Unit 2 Operations Superintendent
T. Cosgrove	Director, Plant Engineering
G. Davie	Training Manager
L. Freeland	Manager, Regulatory Affairs
R. Green	Unit 1 Operations Superintendent
C. Hynes	Operations Training Superintendent
T. Kuhar	LOR Program Administrator
R. Lieb	Manager, System Engineering
D. McBride	System Engineer
E. McFarland	Simulator Supervisor
T. McGourty	System Engineer
R. Mende	Director, Work Management
D. Mickinac	Regulatory Affairs
L. Pearce	Site Vice President
J. Redmond	System Engineer
B. Sepelak	Compliance Supervisor
P. Sena	Manager, Nuclear Operations

NRC Personnel

P. Cataldo	Senior Resident Inspector
T. Fish	Senior Operations Engineer
D. Merzke	Resident Inspector
L. Scholl	Senior Reactor Inspector
G. Smith	Resident Inspector

LIST OF ITEMS, OPENED, CLOSED, AND DISCUSSED

Opened

05000334,412/2004005-02	URI	Acceptability of licensee's LOR written exam development methodology (Section 1R11)

Closed

05000334,412/2003005-01	URI	Acceptability of licensee's simulator testing methodology (Section 4AO5)
05000412/2003003-00	LER	Automatic Reactor Trip Due to Low Steam Generator Water Level (Section 4OA3)
05000334/2003007-00	LER	Inadvertent Reactor Trip During Solid State Protection System Testing (Section 1R14)

Open/Closed

05000334/2004005-01	NCV	Inadequate Design Control Associated With Unit 1 Flood Control Level Switches (Section 1R06)
05000412/2004005-03	NCV	Inadequate Procedural Adherence During the Installation of Scaffolding (Section 1R15)
05000334,412/2004005-04	FIN	Acceptability of Licensee's Simulator Testing Methodology (Section 4AO5)

LIST OF DOCUMENTS REVIEWED

Section 1R4: Equipment Alignment

Procedures

2OM-36.3.B.2, "Valve List - 2EGA," Rev. 12
2OM-36.3.B.3, "Valve List - 2EGF," Rev. 9
2OM-36.3.B.4, "Valve List - 2EGO," Rev. 9
2OM-36.3.B.5, "Valve List - 2EGS," Rev. 10
2OM-36.3.C.8, "Power Supply and Control Switch List - Diesel Generator 2-1," Rev. 9
1OM-13.3.B.2, "Valve List - 1RS," Rev. 7
1OM-13.3.C, "Power Supply and Control Switch List," Rev. 4
2OM-30.3.B.3, "Valve List - 2SWE," Rev. 16
2OM-30.3.C, "Power Supply and Control Switch List," Rev. 15
1OM-21.3.B.1, "Valve List - 1MS," Rev. 14
1OM-21.3.C, "Power Supply and Control Switch List," Rev. 8

Drawings

10080-RM-436-3, "Diesel Starting Air," Rev. 12
10080-RM-436-1, "Diesel Fuel Oil," Rev. 4
10080-RM-436-4A, "Diesel Cooling Water," Rev. 17
10080-RM-436-5A, "Diesel Lube Oil," Rev. 6
8770-RM-413-2, "Containment Depressurization System," Rev. 8
10080-RM-430-1A, "Standby Service Water Supply," Rev. 4
10080-RM-430-1, "Service Water Supply and Distribution," Rev. 27
8700-RM-421-1, "Main Steam," Rev. 15

Miscellaneous Documents

1DBD-21, "Design Basis Document for Main Steam," Rev. 7
System Health Report for Unit 1 Main Steam System

Condition Reports

CR 03-06958

Section 1R05: Fire Protection

Condition Reports

CR-04-06843	Aiming of U-1 App R Lights and App R Drawing Discrepancies
CR-04-06589	Lifting Beam Installed Without Proper Tagging and Bolt Missing
CR-04-06292	Loose Cable Tray Hardware

Section 1R11: Liscensed Operator Requalification Program

Procedures

1/2 –ADM-1359, Simulator Configuration Control, Revision 7

Miscellaneous Documents

Simulator Testing Analysis Action Plan, dated 6/30/04
SGT-5.1, Manual Reactor Trip, Revision 0 (Unit 2)
SGT-5.2, Trip of All Feedwater Pumps Test, Revision 0 (Unit 2)
SGT-5.3, Main Steam Isolation Valves Closure Test, Revision 0 (Unit 2)
SGT-5.4, Complete Loss Of Reactor Coolant Flow Test, Revision 0, (Unit 2)
SGT-5.5, Partial Loss Of Reactor Coolant Flow Test, Revision 0 (Unit 2)
SGT-5.8, Design Basis Accident Loss Of Coolant Accident Transient Test, Revision 0 (Unit 2)
14.1.5.2.2.1, 100% Steady State Test (Unit 2, 2001)
14.1.5.2.2.3, 57% Steady State Test (Unit 2, 2001)
SQT 2.4.13 MSIV Closure, (Unit 1, 2004)
SQT 3.10 Pressurizer Safety Valve Leak HHSI Pumps Inhibited (Unit 1, 2004)
SQT 3.11 Turbine Trip With Rods In Auto (Unit 1, 2004)
SQT 4.138 Pressurizer Steam Space Leak (Unit 1, 2004)
SQT 4.154 Reactor Coolant Pump Trip (Unit 1, 2004)
SQT 4.80 Loss Of DC (Unit 1, 2004)
SQT 4.2 Station Air Compressor Trip (Unit 1, 2004)
List of Condition Reports Related to U1 Simulator for 2002 – 2004
List of Open Simulator Discrepancy Reports
List of BV Unit 1 Simulator Discrepancy Reports Last 2 Years (7/03/02 - 7/12/04)

Condition Reports

CR 04-00994, Unresolved Item (URI) 50-412/03-05-01 RE: Simulator Testing

Section 1R12: Maintenance Rule

Condition Reports

CR-04-05400 FI-1FW-487 (Loop 2 Ch 3) Feed Flow Instrument Multiplier Divider Module Failure

Section 1R13: Maintenance Risk Assessment and Emergent Work Control

Procedures

1/2-ADM-0804, “On-Line Work Management and Risk Assessment,” Rev. 4
1/2-ADM-2033, “Risk Management Program,” Rev. 2
1/2-ADM-2114, “Maintenance Rule Program Administrative Procedure,” Rev.1
Conduct of Operations Procedure 1/2OM-48.1.I, “Technical Specification Compliance,” Rev. 16

NOP-WM-2001, "Work Management Process," Rev. 3

Miscellaneous Documents

10 CFR 50.65(a)(4)

Section 1R19: Post-Maintenance Testing

Work Orders

WO 200076012
WO 200073668
WO 200027470
WO 200075614

Condition Reports

CR 04-05660
CR 04-05675
CR 04-05637

Engineering Change Packages

ECP 03-0428-02

Procedures

1/2CMP-75-MCB-1E, "Testing of Westinghouse and Cutler-Hammer Molded Case Circuit Breakers," Rev. 6

Section 1R20: Refueling and Outage Activities

Condition Reports

CR-04-06292	OMN-16 New Fuel Receipt Procedure Correction
CR-04-06934	Unit 1 New Fuel Receipt Post Job Brief Comments and Possible Enhancements
CR-04-06733	NQA Recommendations Regarding New Fuel Receipt/Inspection Activities

LIST OF ACRONYMS

ACC	Atmospheric Containment Conversion
ADAMS	Agencywide Documents Access and Management System
ADM	Administrative
ADV	Atmospheric Dump Valves
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOP	Abnormal Operating Procedure
ARP	Annunciator Response Procedure
ASP	Alternate Shutdown Panel
BCO	Basis For Continued Operations
BVPS	Beaver Valley Power Station
CB	Control Building
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CR	Condition Report
CV	Cable Vault
DBA	Design Basis Accident
DEP	Drill/Exercise Performance
DG	Diesel Generator
ECP	Engineering Change Package
EDG	Emergency Diesel Generator
ERF	Emergency Response Facility
FB	Fuel Building
FENOC	First Energy Nuclear Operating Company
FIN	Fix-It-Now
FIS	Flow Indicating Switch
HVAC	Heating Ventilation and Air Conditioning
IMC	Inspection Manual Chapter
IPE	Individual Plant Examination
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LOCA	Loss Of Coolant Accident
LOR	Licensed Operator Requalification
LS	Level Switches
MOV	Motor Operated Valve
MR	Maintenance Rule
MS	Main Steam
MSL	Mean Sea Level
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
NUREG	NRC technical report designation
OA	Other Activities
OD	Operability Determination

OM	Operating Manual
OS	Occupational Radiation Safety
OST	Operations Surveillance Test
OWA	Operator Work-Around
PARS	Publicly Available Records
PI	Performance Indicator
PMT	Post-Maintenance Test
PT	Pipe Tunnel
RO	Reactor Operator
RS	Recirculation Spray
RW	River Water
SAT	Systems Approach to Training
SDP	Significance Determination Process
SRO	Senior Reactor Operators
SSC	System, Structure, and Component
SW	Service Water
TDAFW	Turbine-Driven Auxiliary Feedwater
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Issue
V	Volt
VCT	Volume Control Tank
WO	Work Order